WOLF CREEK
NUCLEAR OPERATING CORPORATION

Richard D. Flannigan
Manager Regulatory Affairs

March 31, 2008
RA 08-0027

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Reference:
1) Letter ET 07-0021, dated June 8, 2007, from T. J. Garrett, WCNOC, to USNRC
2) Westinghouse Letter LTR-LIS-08-67, dated March 10, 2008, 10 CFR 50.46 Annual Notification and Reporting for 2007
3) Letter dated March 21, 2008, from USNRC to R. A. Muench, WCNOC

Subject: Docket No. 50-482: 10 CFR 50.46 Annual Report of ECCS Model Changes

Gentlemen:

This letter provides the annual report for the Emergency Core Cooling System (ECCS) Evaluation Model changes and errors for the 2007 model year that affect the Peak Cladding Temperature (PCT) for Wolf Creek Generating Station (WCGS). This letter is provided in accordance with the criteria and reporting requirements of 10 CFR 50.46(a)(3)(ii), as clarified in Section 5.1 of WCAP-13451, "Westinghouse Methodology for Implementation of 10 CFR 50.46 Reporting." Regulation 10 CFR 50.46(a)(3)(ii) states, in part, "For each change to or error discovered in an acceptable evaluation model or in the application of such a model that affects the temperature calculation, the applicant or licensee shall report the nature of the change or error and its estimated effect on the limiting ECCS analysis to the Commission at least annually as specified in section 50.4. If the change or error is significant, the applicant or licensee shall provide this report within 30 days and include with the report a proposed schedule for providing a reanalysis or taking other action as may be needed to show compliance with section 50.46 requirements."

Wolf Creek Nuclear Operating Corporation (WCNOC) has reviewed Reference 2, which addresses 10 CFR 50.46 reporting information pertaining to the ECCS Evaluation Model changes that were implemented by Westinghouse for 2007. The review concludes that with the exception of the LOCBART Pellet Volumetric Heat Generation Rate model change, which was addressed in a 30 day ECCS model change report (Reference 1), the effect of changes to, or errors in, the Evaluation Models on the limiting transient PCT is not significant for 2007. Therefore, the report of the ECCS Evaluation Model changes is provided on an annual basis.

A002
NRR

The Small Break Loss of Coolant Accident (SBLOCA) has been re-analyzed using the 1985 Westinghouse SBLOCA Evaluation Model with NOTRUMP. The re-analysis was performed in support of the Main Steam Isolation Valve (MSIV) and Main Feedwater Isolation Valve (MFIV) Replacement Project to demonstrate conformance with the 10 CFR 50.46 requirements due to the age of the SBLOCA analysis of record (AOR) and the number of peak cladding temperature assessments currently on the SBLOCA PCT summary sheet. The re-analysis accounted for all code modifications as well as plant modifications (except the RCS loose part) since the previous SBLOCA analysis was performed in 1992. The results of the re-analysis were submitted to the NRC for review and approval as part of the License Amendment package for the MSIV/MFIV Replacement Project. The SBLOCA re-analysis was approved by the NRC in NRC Safety Evaluation, letter dated March 21, 2008, from USNRC to R. A. Muench, WCNOC (Amendment 176) (Reference 3) and will replace the current SBLOCA AOR. A summary of the SBLOCA re-analysis and the revised SBLOCA PCT rack-up sheet is include in Attachment III.

Attachment I provides an assessment of the specific changes and enhancements to the Westinghouse Evaluation Models for 2007. Except for the exception noted above with the LOCBART Pellet Volumetric Heat Generation Rate, these model changes and enhancements do not have impacts on the PCT and, generally, will not be presented on the PCT rack-up forms.

Attachment II provides the calculated Large Break Loss of Coolant Accident (LOCA) and Small Break LOCA PCT margin allocations in effect for the 2007 WCGS evaluation models. The PCT values determined in the Small Break and Large Break LOCA analysis of record, combined with all of the PCT allocations, remain well below the 10 CFR 50.46 regulatory limit of 2200 degrees Fahrenheit. Therefore, WCGS is in compliance with 10 CFR 50.46 requirements and no reanalysis or other action is required.

No commitments are identified in this correspondence.

If you have any questions concerning this matter, please contact me at (620) 364-4117, or Diane Hooper at (620) 364-4041.

Sincerely,

Diane M. Hooper for

Richard D. Flannigan

RDF/rlt

Attachment I – Assessment of Changes to the Westinghouse Emergency Core Cooling System (ECCS) Evaluation Models for Large and Small Break Loss of Coolant Accidents (LOCA)

Attachment II – Emergency Core Cooling System (ECCS) Evaluation Model Peak Cladding Temperature (PCT) Margin Utilization

Attachment III - Small Break Loss of Coolant Accident Re-analysis

cc: E. E. Collins (NRC), w/a
V. G. Gaddy (NRC), w/a
B. K. Singal (NRC), w/a
Senior Resident Inspector (NRC), w/a

ASSESSMENT OF CHANGES TO THE WESTINGHOUSE EMERGENCY CORE COOLING SYSTEM (ECCS) EVALUATION MODELS FOR LARGE AND SMALL BREAK LOSS OF COOLANT ACCIDENTS (LOCA)

Non-Discretionary Changes With Peak Cladding Temperature (PCT) Impact

LOCBART Pellet Volumetric Heat Generation Rate

Non-Discretionary Changes With No PCT Impact

BASH-EM Accumulator Water Temperature
BASH Pellet Volumetric Heat Generation Rate
Errors in Reactor Vessel Nozzle Data Collections
LOCBART Specific Heat Model for Optimized Zirlo™ Cladding
Pump Weir Resistance Modeling
NOTRUMP-EM Refined Break Spectrum

Enhancements/Forward-Fit Discretionary Changes

General Code Maintenance (BASH/NOTRUMP)
LOCBART Oxide-to-Metal Ratio

<u>BASH-EM Accumulator Water Temperature</u>
(Non-Discretionary Changes With No PCT Impact)

Background

Reference 1 provided 10 CFR 50.46 reporting information regarding the specification of accumulator water temperature in Appendix K large break LOCA analyses. Reference 1 identified a sensitivity of 1.3°F PCT per 1°F accumulator water temperature that can no longer be supported on a generic basis. Existing analyses have been reviewed to identify any usage of this sensitivity, with plant-specific evaluations completed as required to reflect its elimination. This change represents a Non-Discretionary Change in accordance with Section 4.1.2 of WCAP-13451.

Affected Evaluation Model(s)

1981 Appendix K Large Break LOCA Evaluation Model with BASH

Estimated Effect

The accumulator water temperature modeled in the BASH-EM analysis for Wolf Creek (120°F) is based on the Technical Specification maximum containment average air temperature, so Wolf Creek is unaffected by this issue.

Reference(s)

1. NTD-NRC-95-4409, "1994 Annual Notification of Changes to the Westinghouse Small Break LOCA ECCS Evaluation Model and Large Break LOCA ECCS Evaluation Model, Pursuant to 10 CFR 50.46 (a)(3)(ii)," February 22, 1995.

<u>LOCBART Pellet Volumetric Heat Generation Rate</u>
(Non-Discretionary Changes With PCT Impact)

Background

The LOCBART code has been modified to correct an inverted term in the calculation of the pellet volumetric heat generation rate. This change affects the steady-state and transient heat generation for all three rods and could result in either an increase or decrease in peak cladding temperature for a given calculation. This change represents a Non-Discretionary Change in accordance with Section 4.1.2 of WCAP-13451.

Affected Evaluation Model(s)

1981 Westinghouse Large Break LOCA Evaluation Model with BASH

Estimated Effect

The effect of this change was determined on a plant-by-plant basis. However, due to the vintage of the Wolf Creek analysis-of-record, a plant-specific assessment could not be performed. As such, the LOCBART Pellet Volumetric Heat Generation Rate assessment for Wolf Creek was determined using the bounding value of 45°F from available non-plant-specific sensitivity calculations.

BASH Pellet Volumetric Heat Generation Rate
(Non-Discretionary Changes With No PCT Impact)

Background

The BASH code has been modified to correct an inverted term in the calculation of the pellet volumetric heat generation rate. This change affects the steady-state and transient heat generation for the core average rod prior to bottom-of-core recovery and could result in either an increase or decrease in the cladding temperatures at the beginning of reflood. This change represents a Non-Discretionary Change in accordance with Section 4.1.2 of WCAP-13451.

Affected Evaluation Model(s)

1981 Westinghouse Large Break LOCA Evaluation Model with BASH

Estimated Effect

Sensitivity calculations using BASH and SMUUTH indicated a negligible effect on the core inlet flooding rate during reflood, leading to an estimated impact of 0°F for 10 CFR 50.46 reporting purposes.

Errors in Reactor Vessel Nozzle Data Collections
(Non-Discretionary Changes With No PCT Impact)

Background

Some minor errors were discovered in the reactor vessel nozzle data collections that potentially affect the vessel inlet and outlet nozzle fluid volume, metal mass and surface area. The corrected values have been evaluated for impact on current licensing-basis analysis results and will be incorporated into the plant-specific input databases on a forward-fit basis. These changes represent a closely-related group of Non-Discretionary Changes in accordance with Section 4.1.2 of WCAP-13451.

Affected Evaluation Model(s)

1981 Westinghouse Large Break LOCA Evaluation Model with BASH
1985 Westinghouse Small Break LOCA Evaluation Model with NOTRUMP

Estimated Effect

The differences in the vessel inlet and outlet nozzle fluid volume, metal mass and surface area are relatively minor and would be expected to produce a negligible effect on large break and small break LOCA analysis results, leading to an estimated PCT impact of 0°F for 10 CFR 50.46 reporting purposes.

LOCBART Specific Heat Model for Optimized Zirlo™ Cladding
(Non-Discretionary Changes With No PCT Impact)

Background

An option has been added to the LOCBART code to model the specific heat of Optimized ZIRLO™ cladding. The model is described in the response to Request for Additional Information (RAI) #21 in Section D of Reference 1 and will facilitate compliance with Condition and Limitation #9 of the Safety Evaluation Report for plants with a peak cladding temperature that occurs during blowdown or early reflood. (Note that the extrapolation algorithm described in the RAI response was replaced with an error message and code abort for temperatures below 73°F or above 2400°F.) This change represents a Non-Discretionary Change in accordance with Section 4.1.2 of WCAP-13451.

Affected Evaluation Model(s)

1981 Westinghouse Large Break LOCA Evaluation Model with BASH

Estimated Effect

No domestic plant with a BASH-EM analysis maintained by Westinghouse has both Optimized ZIRLO™ cladding and a peak cladding temperature that occurs during blowdown or early reflood, so there is no impact on any existing analysis results.

Reference(s)

1. WCAP-12610-P-A and CENPD-404-P-A, Addendum 1-A, "Optimized ZIRLO™," July 2006.

Pump Weir Resistance Modeling
(Non-Discretionary Changes With No PCT Impact)

Background

Review of the reactor coolant pump data collections identified instances of either including a weir resistance for a design without a weir or double-counting the weir resistance for a design with a weir. The corrected resistances have been evaluated for impact on existing analysis results and will be incorporated into the plant-specific input databases on a forward-fit basis. This change represents a Non-Discretionary Change in accordance with Section 4.1.2 of WCAP-13451.

Affected Evaluation Model(s)

1981 Westinghouse Large Break LOCA Evaluation Model with BASH

1985 Westinghouse Small Break LOCA Evaluation Model with NOTRUMP

Estimated Effect

Resolving the identified discrepancies has been evaluated as having a negligible effect on existing results, leading to an estimated PCT impact of 0°F for 10 CFR 50.46 reporting purposes.

NOTRUMP-EM Refined Break Spectrum
(Non-Discretionary Changes With No PCT Impact)

Background

During the course of reviewing several extended power uprate and replacement steam generator Small Break LOCA (SBLOCA) analyses, the Nuclear Regulatory Commission (NRC) questioned the break spectrum analyzed in the NOTRUMP evaluation model (EM). The NRC was concerned that the resolution of the break spectrum used in the NOTRUMP EM (1.5, 2, 3, 4, and 6 inch cases) may not be fine enough to capture the worst break with regard to limiting peak clad temperature as per 10 CFR 50.46. That is, the plant could be SBLOCA limited with regard to overall LOCA results.

In response to this, Westinghouse performed some preliminary work indicating that in some cases more limiting results could be obtained from non-integer break sizes; however, the magnitude of the impact was far less than that shown in preliminary work performed by the NRC. Based on this, Westinghouse performed evaluations to determine if all currently operating plants would maintain compliance with the 10 CFR 50.46 acceptance criteria when considering a refined SBLOCA break spectrum. It should be noted that use of a refined break spectrum is not an error, but a change, since evaluating only integer break sizes has been the standard practice since the initial licensing of NOTRUMP.

This change represents a Non-Discretionary Change in accordance with Section 4.1.2 of WCAP-13451.

Affected Evaluation Model(s)

1985 Westinghouse Small Break LOCA Evaluation Model with NOTRUMP

Estimated Effect

Consistent with the method described in Reference 1, for plants with low SBLOCA peak cladding temperatures (i.e., less than 1700°F) and overall SBLOCA results that are significantly non-limiting when compared with large break LOCA (LBLOCA) results, no explicit refined break spectrum calculations were performed, leading to an estimated impact of 0°F for 10 CFR 50.46 reporting purposes. For plants with high SBLOCA PCTs (i.e., equal to or greater than 1700°F), explicit refined break spectrum calculations were performed, and PCT penalties were assessed, if necessary.

Reference(s)

1. LTR-NRC-06-44, "Transmittal of LTR-NRC-06-44 NP-Attachment, 'Response to NRC Request for Additional Information on the Analyzed Break Spectrum for the Small Break Loss of Coolant Accident (SBLOCA) NOTRUMP Evaluation Model (NOTRUMP EM), Revision 1,' (Non-Proprietary)," July 14, 2006.

General Code Maintenance
(Discretionary Changes)

Background

Various changes have been made to enhance the usability of the codes and to help preclude errors in analyses. This includes items such as modifying input variable definitions, units, and defaults, improving the input diagnostic checks, enhancing the code output, optimizing active coding, and eliminating inactive coding. These changes represent Discretionary Changes that will be implemented on a forward-fit basis in accordance with Section 4.1.1 of WCAP-13451.

Affected Evaluation Model(s)

1981 Westinghouse Large Break LOCA Evaluation Model with BASH
1985 Westinghouse Small Break LOCA Evaluation Model with NOTRUMP

Estimated Effect

The nature of these changes leads to an estimated PCT impact of 0°F.

LOCBART Oxide-to-Metal Ratio
(Discretionary Changes)

Background

An option has been added to the LOCBART code to convert the user-specified zirconium-oxide thickness to equivalent cladding reacted. This adjustment is made during problem initialization, and the cladding outside diameter is modified accordingly. This change represents a Discretionary Change that will be implemented on a forward-fit basis in accordance with Section 4.1.1 of WCAP-13451.

Affected Evaluation Model(s)

1981 Westinghouse Large Break LOCA Evaluation Model with BASH

Estimated Effect

This change is expected to produce a minimal effect on the limiting peak cladding temperature, leading to an estimated effect of 0°F.

LOCA Evaluation of ENUSA Fuel Assemblies for the Wolf Creek (SAP) Security of Supply Program

Background

As discussed in Reference 1, to ensure the security of supply of fuel assemblies to Wolf Creek, Wolf Creek Nuclear Operating Company has requested that ENUSA provide four Non-IFBA assemblies for Cycle 17. As a result, there will be some changes in the fuel parameters from what was considered in the current Wolf Creek LOCA analyses (fuel manufacturing by Westinghouse-Columbia (W-C)). Westinghouse is being requested to confirm that the current LOCA analyses for Wolf Creek remain bounding for the ENUSA assemblies.

Evaluation

For Large Break and Small Break LOCA, a comparison of the fuel features for the current Westinghouse fuel (as analyzed in the current LOCA analyses) and the ENUSA fuel was performed. Based on the comparison, the only difference between the W-C and the ENUSA fuel is the pellet fabrication process which impacts pellet densification. The rod internal pressure and fuel temperatures were compared and the small differences found will result in a negligible impact on LBLOCA and SBLOCA peak cladding temperature with respect to the four ENUSA fuel assemblies.

LOCA Forces and Post-LOCA do not model the fuel assemblies in sufficient detail. ENUSA fuel is only being placed in a small number of assemblies relative to the entire core; therefore, LOCA Forces and Post-LOCA are not impacted by the four ENUSA assemblies.

Estimated Effect

The ENUSA Non-IFBA fuel assemblies are acceptable for use in the Wolf Creek Cycle 17 core with respect to the LBLOCA, SBLOCA, LOCA Forces, and Post-LOCA analyses. Only large break and small break LOCA have sufficient modeling detail to be affected by the differences between the proposed ENUSA fuel assemblies and the resident fuel. This evaluation shows a negligible PCT impact for LBLOCA and SBLOCA with respect to the four ENUSA fuel assemblies.

References

1. NF-SAP-07-1, "Preliminary Reload Design Initialization Checklist (RDIC) for Cycle 17," January 8, 2007.

EMERGENCY CORE COOLING SYSTEM (ECCS) EVALUATION MODEL PEAK CLADDING TEMPERATURE (PCT) MARGIN UTILIZATION

*** LARGE BREAK LOCA PEAK CLAD TEMPERATURE (PCT) MARGIN UTILIZATION ***

Evaluation Model:	1981 EM with BASH
Fuel:	17x17 V5H w/IFM, non-IFBA, 275 psig
Peaking Factor:	FQ=2.50, FdH=1.65
SG Tube Plugging:	10%
Power Level:	3565 MWth
Limiting transient:	Cd=0.4, Min. SI, Reduced Tavg

		Clad Temp (°F)	Ref.	Notes
LICENSING BASIS				
Analysis of Record PCT		**1916°F**	**1**	**(a)**
MARGIN ALLOCATIONS (ΔPCT)				
A. PRIOR PERMANENT ECCS MODEL ASSESSMENTS				
1.	Structural Metal Heat Modeling	-25	8	
2.	LUCIFER Error Corrections	-6	10	
3.	Skewed Power Shape Penalty	152	11	
4.	Hot Leg Nozzle Gap Benefit	-136	11	
5.	SATAN-LOCTA Fluid Error	15	2	
6.	LOCBART Spacer Grid Single-Phase Heat Transfer Error	15	9	
7.	LOCBART Vapor Film Flow Regime Heat Transfer Error	9	12	
8.	LOCBART Cladding Emissivity Errors	6	13	
9.	LOCBART Radiation to Liquid Logic Error Correction	17	14	
B. PLANNED PLANT CHANGE EVALUATIONS				
1.	Loose Parts Evaluation	20	3	
2.	Effects of Containment Purging	0	4	
3.	Cycle 10 Fuel Assembly Design Changes	95	5	
4.	Fuel Rod Crud	0	6	
C. 2007 PERMANENT ECCS MODEL ASSESSMENTS				
1.	LOCBART Pellet Volumetric Heat Generation Rate	45	15	
D. TEMPORARY ECCS MODEL ISSUES		0		
E. OTHER				
1.	Cold Leg Streaming Temperature Gradient	0	8	(b)
2.	Rebaseline of AOR (12/96)	-63	9	(c)
3.	LOCBART Zirc-Water Oxidation Error	28	7	(d)

LICENSING BASIS PCT + MARGIN ALLOCATIONS **PCT = 2088°F**

CUMULATIVE ABSOLUTE MAGNITUDE OF PCT CHANGES SINCE LAST 30-DAY REPORT (LETTER ET 07-0021) **$\Sigma|\Delta PCT|$ = 0°F**

References:

1. Westinghouse Topical Report WCAP-13456, "Wolf Creek Generating Station NSSS Rerating Licensing Report," October 1992.
2. Westinghouse to WCNOC letter SAP-97-102, "Wolf Creek Nuclear Operating Corporation, Wolf Creek Generating Station, 10 CFR 50.46 Annual Notification and Reporting," February 17, 1997.
3. Westinghouse to WCNOC letter SAP-90-148, "Wolf Creek Nuclear Operating Corporation, RCS Loose Parts Evaluation," April 18, 1998.
4. Westinghouse to WCNOC letter SAP-94-102, "Containment Mini Purge Isolation Valve Stroke Time Increase," January 12, 1994.
5. Westinghouse to WCNOC letter 97SAP-G-0009, "Wolf Creek Nuclear Operating Corporation, Wolf Creek Generating Station, Safety Assessment for the Wolf Creek Generating Station with ZIRLO™ Fuel Assemblies," February 7, 1997.
6. Westinghouse to WCNOC letter 97SAP-G-0075, "Wolf Creek Nuclear Operating Corporation, Wolf Creek Generating Station, Wolf Creek Crud Deposition/Axial Offset Anomaly Safety Evaluation," September 29, 1997.
7. Westinghouse to WCNOC letter 00SAP-G-0006, "Wolf Creek Nuclear Operating Corporation, Wolf Creek Generating Station, Wolf Creek Cycle 12 LOCA Current Limits," February 10, 2000.
8. Westinghouse to WCNOC letter SAP-93-701, "Wolf Creek Nuclear Operating Corporation, Wolf Creek Generating Station, 10 CFR 50.46 Notification and Reporting Information," January 25, 1993.
9. Westinghouse to WCNOC letter SAP-99-148, "Wolf Creek Nuclear Operating Corporation, Wolf Creek Generating Station, 10 CFR 50.46 BART/BASH Evaluation Model Mid-Year Notification and Reporting for 1999," September 22, 1999.
10. Westinghouse to WCNOC letter SAP-94-703, "Wolf Creek Nuclear Operating Corporation, Wolf Creek Generating Station, 10 CFR 50.46 Notification and Reporting," February 8, 1994.
11. Westinghouse to WCNOC letter SAP-95-716, "Wolf Creek Nuclear Operating Corporation, Wolf Creek Generating Station, LOCA Axial Power Shape Sensitivity Model," August 14, 1995.
12. Westinghouse to WCNOC letter SAP-00-118, "Wolf Creek Nuclear Operating Corporation, Wolf Creek Generating Station, 10 CFR 50.46 Appendix K (BART/BASH/NOTRUMP) Evaluation Model, Mid-Year Notification and Reporting for 2000," June 30, 2000.
13. Westinghouse to WCNOC letter SAP-00-150, "Wolf Creek Nuclear Operating Corporation, Wolf Creek Generating Station, 10 CFR 50.46 BART/BASH Evaluation Model Mid-Year Notification and Reporting for 2000," December 2000.
14. Westinghouse to WCNOC letter SAP-02-32, "10 CFR 50.46 BART/BASH Evaluation Model Mid-Year Notification and Reporting for 2002," June 2002.
15. LTR-LIS-07-312, "10 CFR 50.46 Reporting Text for LOCBART Version 37.0 Issues and Revised PCT Rackup Sheets for Wolf Creek," May 2007.

Notes:

(a) An evaluation was performed to support removal of the transition core penalty for Cycle 12 (Ref. 7).

(b) A PCT benefit of < 2.5°F was assessed, however, a benefit of 0°F will be tracked for reporting purposes.

(c) This previously unclaimed benefit was realized through prior rebaseline of the limiting case.

(d) This assessment is a function of analysis PCT plus certain margin allocations and as such may increase/decrease with margin allocation changes.

*** SMALL BREAK LOCA PEAK CLAD TEMPERATURE (PCT) MARGIN UTILIZATION ***

Evaluation Model:	1985 EM with NOTRUMP
Fuel:	17x17 V5H w/IFM, non-IFBA, 275 psig
Peaking Factor:	FQ=2.50, FdH=1.65
SG Tube Plugging:	10%
Power Level:	3565 MWth
Limiting transient:	3-inch Break

LICENSING BASIS

	Clad Temp (°F)	Ref.	Notes
Analysis of Record PCT	**1510**	**1**	
MARGIN ALLOCATIONS (ΔPCT)			
A. PRIOR PERMANENT ECCS MODEL ASSESSMENTS			
1. Effect of SI in Broken Loop	150	10	
2. Effect of Improved Condensation Model	-150	10	
3. Drift Flux Flow Regime Errors	-13	11	
4. LUCIFER Error Corrections	-16	11	
5. Boiling Heat Transfer Correlation Error	-6	12	
6. Steam Line Isolation Logic Error	18	12	
7. Axial Nodalization, RIP Model Revision and SBLOCTA Error Corrections Analysis	26	13	
8. NOTRUMP Specific Enthalpy Error	20	2	
9. SBLOCTA Fuel Rod Initialization Error	2	14	
10. NOTRUMP Mixture Level TrackinglRegion Depletion Errors	13	15	
11. NOTRUMP Bubble Rise/Drift Flux Model Inconsistency Corrections	0	16	
B. PLANNED PLANT CHANGE EVALUATIONS			
1. Loose Part Evaluation	45	3	
2. Cycle 10 Fuel Assembly Design Change	1	6	
3. Reduced Feedwater Inlet Temperature	10	4	
4. Fuel Rod Crud	4	5	(a)
5. Auxiliary Feedwater Temperature Increase	16	8,9	(b)
6. High Head SI Flow Reduction	35	17	
C. 2007 PERMANENT ECCS MODEL ASSESSMENTS			
1. None	0		
D. TEMPORARY ECCS MODEL ISSUES	0		
E. OTHER			
1. Cold Leg Streaming Temperature Gradient	7	7	

LICENSING BASIS PCT + MARGIN ALLOCATIONS **PCT = 1672°F**

CUMULATIVE ABSOLUTE MAGNITUDE OF PCT CHANGES SINCE LAST 30-DAY REPORT (LETTER ET 99-0024) **$\Sigma|\Delta PCT|$ = 35°F**

References:

1. Westinghouse Topical Report WCAP-13456, "Wolf Creek Generating Station NSSS Rerating Licensing Report," October 1992.
2. Westinghouse to WCNOC letter SAP-96-705, "Wolf Creek Nuclear Operating Corporation, Wolf Creek Generating Station, 10 CFR 50.46 Notification and Reporting," February 9, 1996.
3. Westinghouse to WCNOC letter SAP-90-148, "Wolf Creek Nuclear Operating Corporation, RCS Loose Parts Evaluation," April 18, 1990.
4. Westinghouse to WCNOC letter SAP-96-119, "Wolf Creek Nuclear Operating Corporation, Wolf Creek Generating Station, Small Break LOCA Evaluation for Reduced Feedwater Temperature," May 30,1996.
5. Westinghouse to WCNOC letter 97SAP-G-0075, "Wolf Creek Nuclear Operating Corporation, Wolf Creek Generating Station, Wolf Creek Crud Deposition/Axial Offset Anomaly Safety Evaluation," September 29, 1997. (This penalty will be carried until such time it is determined to no longer apply).
6. Westinghouse to WCNOC letter 97SAP-G-0009, "Wolf Creek Nuclear Operating Corporation, Wolf Creek Generating Station, Safety Assessment for the Wolf Creek Generating Station with ZIRLO™ Fuel Assemblies," February 7, 1997.
7. Westinghouse to WCNOC letter SAP-93-701, "Wolf Creek Nuclear Operating Corporation, Wolf Creek Generating Station, 10 CFR 50.46 Notification and Reporting Information," January 25, 1993.
8. Westinghouse to WCNOC letter SAP-98-138, "Wolf Creek Nuclear Operating Corporation, Wolf Creek Generating Station, Assessment of an Increase in Auxiliary Feedwater Temperature," July 23, 1998.
9. Westinghouse to WCNOC letter 00SAP-G-0006, "Wolf Creek Nuclear Operating Corporation, Wolf Creek Generating Station, Wolf Creek Cycle 12 LOCA Current Limits," February 10, 2000.
10. Westinghouse to WCNOC letter SAP-93-718, "Wolf Creek Nuclear Operating Corporation, Wolf Creek Generating Station, Safety Injection in the Broken Loop," September 22, 1993.
11. Westinghouse to WCNOC letter SAP-94-703, "Wolf Creek Nuclear Operating Corporation, Wolf Creek Generating Station, 10 CFR 50.46 Notification and Reporting," February 8, 1994.
12. Westinghouse to WCNOC letter SAP-94-722, "Wolf Creek Nuclear Operating Corporation, Wolf Creek Generating Station, 10 CFR 50.46 Notification and Reporting," August 18, 1994.
13. Westinghouse to WCNOC letter SAP-94-727, "Wolf Creek Nuclear Operating Corporation, Wolf Creek Generating Station, SBLOCTA Axial Nodalization," October 27, 1994.
14. Westinghouse to WCNOC letter SAP-97-102, "Wolf Creek Nuclear Operating Corporation, Wolf Creek Generating Station, 10 CFR 50.46 Annual Notification and Reporting," February 17, 1997.
15. Westinghouse to WCNOC letter SAP-00-118, "Wolf Creek Nuclear Operating Corporation, Wolf Creek Generating Station, 10 CFR 50.46 Appendix K (BART/BASH/NOTRUMP) Evaluation Model, Mid-Year Notification and Reporting for 2000," June 30, 2000.
16. Westinghouse to WCNOC letter SAP-03-33, "10 CFR 50.46 Mid-Year Notification and Reporting for 2003," November 14, 2003.
17. Westinghouse to WCNOC letter SAP-04-33, "Wolf Creek Nuclear Operating Corporation, Wolf Creek Generating Station, High Head Safety Injection Flow Rate Reduction -Final Evaluation," June 11,2004.

Notes:

(a) This penalty will be carried until such time it is determined to no longer apply.

(b) This increase in auxiliary feedwater temperature was originally evaluated in Reference 8 as a 16°F penalty. However, this change was not implemented until the Cycle 12 reload. Reference 9 represents the transmittal of the Cycle 12 LOCA Reload Current Limits.

Small Break Loss of Coolant Accident Re-analysis

Background

The Small Break Loss of Coolant Accident (SBLOCA) has been re-analyzed using the 1985 Westinghouse SBLOCA Evaluation Model (EM) with NOTRUMP (References 1, 2, and 3). The re-analysis was performed in support of the Main Steam Isolation Valve (MSIV)/Main Feedwater Isolation Valve (MFIV) Replacement Project to demonstrate conformance with the 10 CFR 50.46 requirements with consideration of the age (1992) of the SBLOCA analysis of record (AOR) and also the number of peak cladding temperature (PCT) assessments currently tracked on the SBLOCA PCT summary sheet. NOTE: Several updates have been made to the NOTRUMP-EM since the previous AOR was completed, including the use of the COSI condensation model and Safety Injection (SI) in the broken loop (Reference 3), which have been incorporated in this analysis. The analysis accounted for all code modifications as well as plant modifications (excluding the RCS loose part) since the previous SBLOCA analysis was performed in 1992. The PCT penalty due to the RCS loose part evaluation is retained as a line item, since the loose part has not been recovered from the RCS.

The analysis considered a spectrum of cold leg breaks of equivalent diameters of 2, 3, 4 and 6 inches as well as an 8.75 inch diameter accumulator line break.

Affected Evaluation Model(s)

1985 Westinghouse Appendix K Small Break LOCA Evaluation Model with NOTRUMP.

Results

The review of the Wolf Creek SBLOCA analysis results show that the limiting peak cladding temperature of 936°F occurs for the 4-inch break at beginning of life (BOL). The revised SBLOCA PCT rack-up sheet is provided on the next page.

References:

1. WCAP-10079-P-A, “NOTRUMP - A Nodal Transient Small Break and General Network Code,” Meyer, P. E., August 1985.

2. WCAP-10054-P-A, “Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code,” Lee, N., et al., August 1985.

3. WCAP-10054-P-A, Addendum 2, Revision 1, “Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model,” Thompson, C. M., et al., July 1997.

4. NRC Safety Evaluation in Letter dated March 21, 2008, from USNRC to R. A. Muench, WCNOC (Amendment 176)

***SMALL BREAK LOCA PEAK CLAD TEMPERATURE (PCT) MARGIN UTILIZATION ***

Evaluation Model:	1985 EM with NOTRUMP
Fuel:	17x17 RFA-2 w/IFM
Peaking Factor:	FQ=2.50, FdH=1.65
SG Tube Plugging:	10%
Power Level:	3565 MWth
Limiting transient:	4-inch Break

	Clad Temp (°F)	Ref.	Notes
LICENSING BASIS			
Analysis of Record PCT	**936**	**1**	
MARGIN ALLOCATIONS (ΔPCT)			
A. PRIOR PERMANENT ECCS MODEL ASSESSMENTS			
12. None	0		
B. PLANNED PLANT CHANGE EVALUATIONS			
7. Loose Part Evaluation	45	2	(a)
C. 2007 PERMANENT ECCS MODEL ASSESSMENTS			
1. None	0		
D. TEMPORARY ECCS MODEL ISSUES	0		
1. None			
E. OTHER			
1. None	0		
LICENSING BASIS PCT + MARGIN ALLOCATIONS	**PCT = 981°F**		

References:

1. WCAP-16717-P, Rev. 0, "Wolf Creek Generating Station (SAP), MSIV/MFIV Replacement Project, Small Break Loss of Coolant Accident Analysis Engineering Report," January 2007.

2. SAP-90-148/NS-OPLS-OPL-I-90-239, "Wolf Creek Nuclear Operating Corporation, RCS Loose Part Evaluation," April 1990.

Notes:

(a) This penalty will be carried to track the loose part which has not been recovered.